Westinghouse Non-Proprietary Class 3

WCAP-14762

AP600 Reactor Internals Flow-Induced Vibration Assessment Program

Westinghouse Energy Systems

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14762

AP600 REACTOR INTERNALS

FLOW-INDUCED VIBRATION ASSESSMENT PROGRAM

March, 1996

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Energy Systems Business Unit
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1.0 INTRODUCTION

With respect to the reactor internals preoperational test program, the first AP600 plant reactor vessel internals are classified as prototype as defined in Regulatory Guide 1.20, Rev 2. AP600 reactor vessel internals do not represent a first-of-a-kind or unique design based on the arrangement, design, size, or operating conditions. The units referenced as supporting the AP600 reactor vessel internals design features and configuration have successfully completed vibration assessment programs including vibration measurement programs. These units have subsequently demonstrated extended satisfactory inservice operation.

The vibration assessment approach is believed to meet the intent of Regulatory Guide 1.20 and is similar to the approach taken by Westinghouse on previous plants. Additional background on the Westinghouse position with regard to this guide is provided in Westinghouse Nuclear Safety Position Papers (References 15 and 16).

The purpose of this AP600 reactor internals vibration assessment program is to demonstrate structural adequacy with respect to flow-induced pump vibrations. Estimates of flow-induced vibration levels and forces (or relative values) of the AP600 plant are made on the basis of scale model tests, tests on prototype reactors and the results of analytical calculations. Based on this information, the vibratory behavior of the reactor internals is well characterized, and the vibration amplitudes are sufficiently low for structural adequacy of the components.

The H. B. Robinson No. 2 plant has been established as the prototype design for 3 loop plant internals and was instrumented and tested during Hot Functional testing. The test and analysis results of the 3 loop configuration of Reference (1) demonstrate that the vibration levels of the reactor internals components are low and that the vibrations are adequately characterized to assure structural integrity. These results are further augmented by References 2, 3, 4, and 5 to address the effects of successive hardware improvements in Westinghouse designs which are discussed in the following sections.

The AP600 reactor internals are generally similar to subsequent 3 loop, 12 and 14 foot core designs (specifically Doel 3 and Doel 4) which have incorporated these improvements, and on which instrumented plant test programs have been completed. The dimensions of the AP600 core barrel and reactor vessel to core barrel downcomer annulus are similar to those of Doel 3 and Doel 4. The AP600 guide tube and support column designs are the same (with minor differences which are to be discussed later) as the designs used in Doel 3 and Doel 4. The upper internals components vibration responses were measured at Doel 3 (Reference 10) and the lower internals were measured at Doel 4 (Reference 9).

The total reactor flow rates in these instrumented tests were approximately [ ]^{b} gpm and [ ]^{b} gpm as compared to the AP600 value of [ ]^{b} gpm ([ ]^{b} gpm during Hot Functional testing). As a consequences of the lower flowrate in the AP600 design, the flow velocities in the AP600 reactor are significantly less than in the previously instrumented 3 loop plants.

Table 1.1 lists the ratios of the AP600 (Reference 13) to Doel 4 (Reference 14) plant velocities at various locations based on the above flow rates. The lower velocities in the AP600 design result in lower turbulence excitation forces for flow-induced vibration in the AP600 reactor internals than in the reference units.

The AP600 internals have two component changes:

1) The baffle plates and formers have been replaced by a reflector which rests on the lower core support plate.

2) The structures below the lower core support plate have been modified consistent with the routing of the in-core instrumentation cables through the upper head and to accommodate a larger plate to suppress vortices in the core inlet plenum.

The pre-operational test program of the first AP600 plant includes a vibration measurement program and a pre- and post-hot functional inspection program. This program satisfies the guidelines for a Regulatory Guide 1.20 Prototype Category plant. The AP600 reactor internals design does not require supplemental testing including component vibration tests, flow tests, or scale model tests. AP600 plants subsequent to the first plant will be subject to the pre- and post-hot functional inspection program. The program for plants subsequent to the first plant satisfies the guidelines for a Non-Prototype Category IV plant.

TABLE 1.1

REACTOR INTERNALS COOLANT FLOW VELOCITY RATIOS

Location	AP600/Doel 4
Inlet Nozzle	0.772
Downcomer (Reactor Vessel To Core Barrel Annulus)	0.570
Lower Plenum	0.667
Reactor Core	0.683
Outlet Nozzle	0.845

2.0 SUMMARY

The AP600 reactor internals represent an evolutionary change from the original Westinghouse designs for the prototype 2, 3 and 4 loop plants.

The AP600 reactor and internals are dimensionally similar to previous Westinghouse 3 loop plants and therefore the vibratory behavior of its major structural components will be similar. It should be noted that although similar in size to previous Westinghouse 3 loop plants, the flow rate in the AP600 is significantly lower. A comparison of flow rates for various plants is included in Section 4.0 which shows the AP600 flow rate to be approximately 21 percent lower than the H. B. Robinson flowrate and approximately 30% below more recent 3-loop plants.

The AP600 reactor internals incorporate many design features that were introduced in other Westinghouse plants. The inverted top hat (or UHI-style) upper internals was instrumented at Sequoyah Unit 1, the lead UHI (Upper Head Injection) 4 loop plant. This style upper internals with it's 17x17 guide tubes and solid, rather than slotted, upper support columns has become the standard design for subsequent 3 and 4 loop plants. These features were also incorporated into the replacement 2 loop upper internals assemblies provided for Prairie Island Units 1 and 2.

The lower internals has also undergone changes. The thermal shield or neutron panels have been removed from the core barrel which results in a decreased flow velocity in the reactor vessel to core barrel downcomer annulus. The original lower core and support plates have been combined into a single plate which was introduced in the 3 and 4 loop XL (EXTENDED LENGTH fuel) plants. These lower internals were instrumented in preoperational vibration measurement programs at Doel 4 and Paluel 1.

The evolutionary changes that were introduced in previous Westinghouse plants have demonstrated their adequacy through a combination of plant measurement programs and successful operation over several fuel cycles. In addition, all Westinghouse units have undergone a thorough inspection of the reactor vessel internals before and after Hot Functional Testing. These inspections have not indicated any abnormal wear, impacting or excessive vibration displacements. The assessment in this report indicates that flow-induced vibrations of the AP600 internals are adequately low and that the vibrations in the dominant (beam) modes are lower than in standard plants.

The changes that are unique to the AP600 reactor internals will be instrumented for a Preoperational Vibration Measurement Program which is discussed in Section 8.0. The results of the prototype AP600 test program are expected to further demonstrate the adequacy of the reactor internals design.

3.0 DESIGN DIFFERENCES AND RELATIONSHIP TO FLOW-INDUCED VIBRATIONS

As mentioned earlier, the Westinghouse 3 loop prototype plant is H. B. Robinson. Subsequent structural differences include modifications resulting from the use of 17x17 fuel, the removal of the annular thermal shield and the change to the (UHI style) inverted top hat support structure configuration. The effects of these changes were determined by instrumented test programs at Trojan, Sequoyah 1, Doel 3 and Doel 4. The primary changes from the Doel 3 and Doel 4 designs are the addition of a vortex suppression plate beneath the lower core plate, the inclusion of a radial reflector within the core barrel, and the change from bottom to top mounted in-core instrumentation. The design similarities, and these design differences, and their relationships to the flow-induced vibrational characteristics are discussed in the following sections.

3.1 Lower Internals Assembly

The AP600 lower internals are similar to the 3XL design (Figure 3.1) for which a pre-operational vibration measurement program was conducted at Doel 4. The diameter, length and thickness of the core barrel structure are almost identical between these two plants. The 3XL and AP600 lower internals both have the single combined lower core support plate.

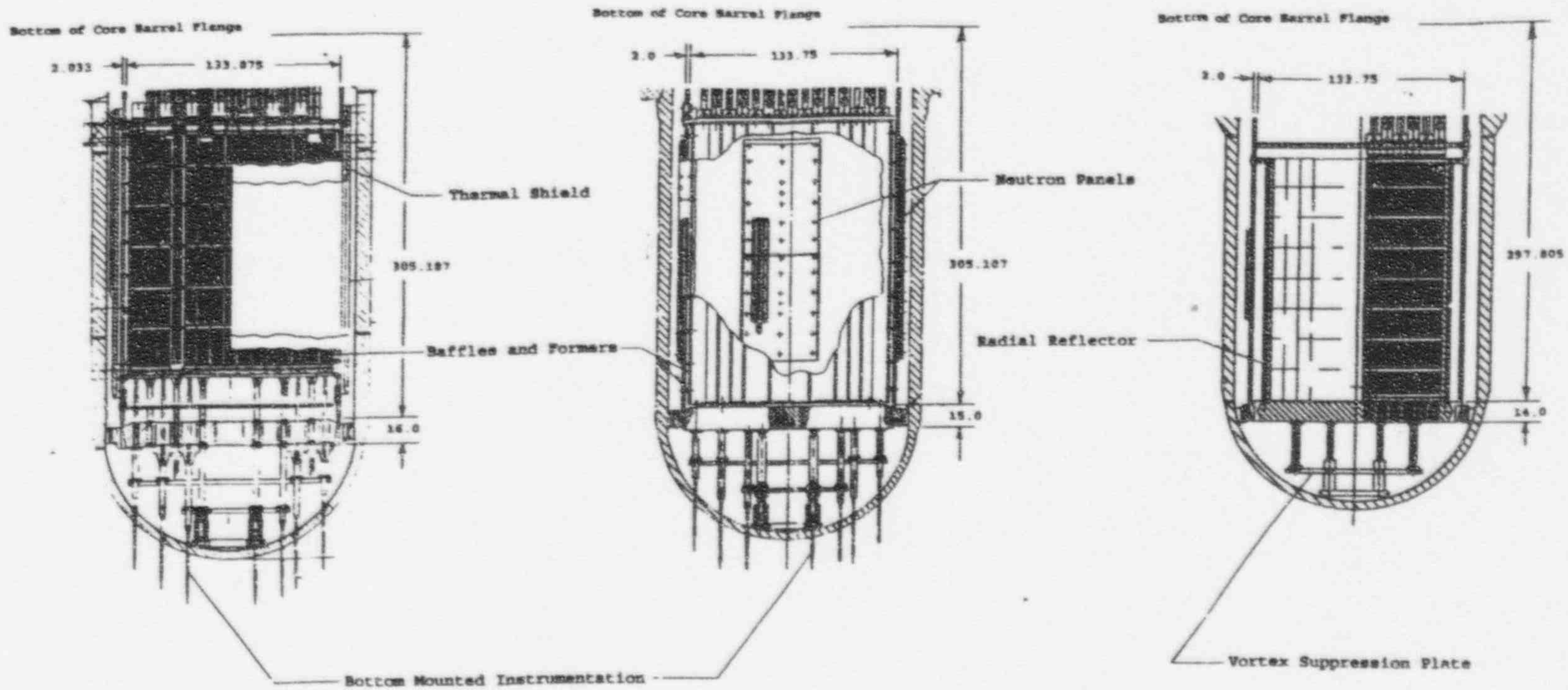
Differences between the AP600 and the referenced designs include the removal of the thermal shield or neutron pads, replacement of the baffle former assembly with the reflector, elimination of the guide columns for the incore instrumentation and addition of a vortex suppression plate.

Results from Indian Point and Trojan (Reference 2), as well as scale model test results (Reference 3), show that core barrel vibration of plants with neutron shielding pads is less than that of plants with thermal shields. The removal of the neutron pads in the AP600 design furtner reduces flow velocities in the reactor vessel core barrel downcomer annulus by increasing the flow area. In addition, core barrel responses were measured in a plant vibration measurement program at Paluel (Reference 11), which is a 4 loop plant without neutron pads. These responses are lower than those for 4 loop neutron pad core barrels measured at Trojan 1 for the same flow rate. These factors and the lower flow rate of the AP600 design tend to result in a lower beam mode response of the AP600 core barrel than the core barrel responses measured at Doel 4.

Removal of the baffle formers reduces the core barrel shell and beam mode stiffness. The effect on beam modes is small relative to the effect of the downcomer velocity difference. The reduced stiffness combined with a potential for higher added water mass in the barrel-reflector annulus, however, will tend to increase the vibratory response levels of the core barrel shell modes relative to the reference plant while the reduced velocity will tend to reduce them. The AP600 core barrel shell mode responses are expected to be higher than in the reference plant (Section 7) but to result in acceptable stress levels.

The reflector is a cylinder that rests on and is bolted to the lower core support plate. Radial restraint of the upper end of the reflector is provided by 4 pins that are attached to the core barrel.

The change to top-mounted in-core instrumentation has simplified the structure beneath the lower support plate. Although the secondary core support (energy absorber) has been retained, the guide columns and tie plates associated with the bottom mounted in-core instrumentation have been eliminated.



H. B. Robinson No. 2 Doel 4/Doel 3 AP600

Figure 3.1 Design Differences In Lower Internals

A vortex suppression plate (Figure 3.2) has been added. This plate consists of two annular rings connected by four spokes. The inner ring of the vortex suppression plate is supported by four columns which also form part of the secondary core support. The outer annular ring is suspended beneath the lower core plate by eight equally spaced butt columns. The vortex suppression plate has been shown to suppress the formation of standing vortices in the core inlet plenum in laboratory testing (Reference 29). The response of this structure will be measured during the AP600 preoperational test.

3.2 Upper Internals Assembly

Figure 3.3 show the upper internals for the 3 loop prototype, Doel 3/Doel 4 and AP600. The upper support plate for the guide tubes and support columns becomes a simple structure in the form of an inverted top hat as shown in Figure 3.4. It consists of a flange, a skirt and a thick base plate connected by two circumferential welds. The original design included a thinner plate stiffened by a system of ribs attached by corner-welding (deepbeam structure). The change to the new design results in shorter and stronger (17x17 AS) guide tubes which are less susceptible to flow-induced vibrations.

The AP600 inverted top hat (UHI style) upper support structure is the same design as in the Doel 3 and Doel 4 upper internals. Therefore, the same general vibration behavior is expected. The components of the upper internals are excited by turbulent forces due to axial and cross-flow in the upper plenum and by pump related excitations (References 2 and 5).

The upper support assembly skirt (Figure 3.4) is approximately 10 inches longer than the previous 3 loop internals inverted top hat designs such as Doel 3 and Doel 4. However, the structural characteristics are not expected to change significantly relative to those observed at Doel 3 and Doel 4 since the support plate, flange and skirt thicknesses are the same.

The AP600 upper internals assembly has different numbers of guide tubes and support columns as compared to the Doel 3 and Doel 4 plants (Figure 3.5). The AP600 design has a total of 61 guide tubes versus 53 and 57 in Doel 3 and Doel 4 respectively. In addition, the AP600 design has 42 support columns rather than the 40 used in the Doel 3 and Doel 4 plants. These differences are expected to have little effect on the upper internals responses that were measured in the 3XL 1/7 scale model and Doel 3 plant tests. Since the guide tubes and support columns respond as individual beams restrained by the upper support plate and upper core plate at their top and bottom ends respectively, additional numbers of these components will not alter the dynamic characteristics of these components. Mean loads and vibratory amplitudes are calculated for the AP600 array using a calculation method that is based on flow test data.

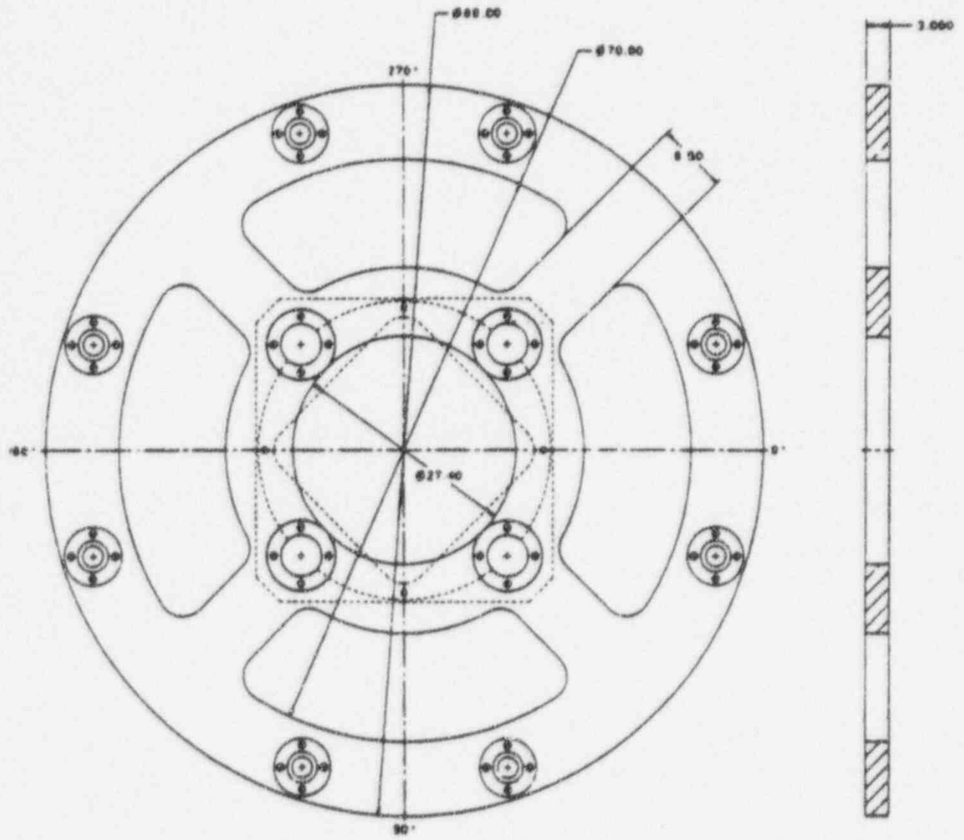
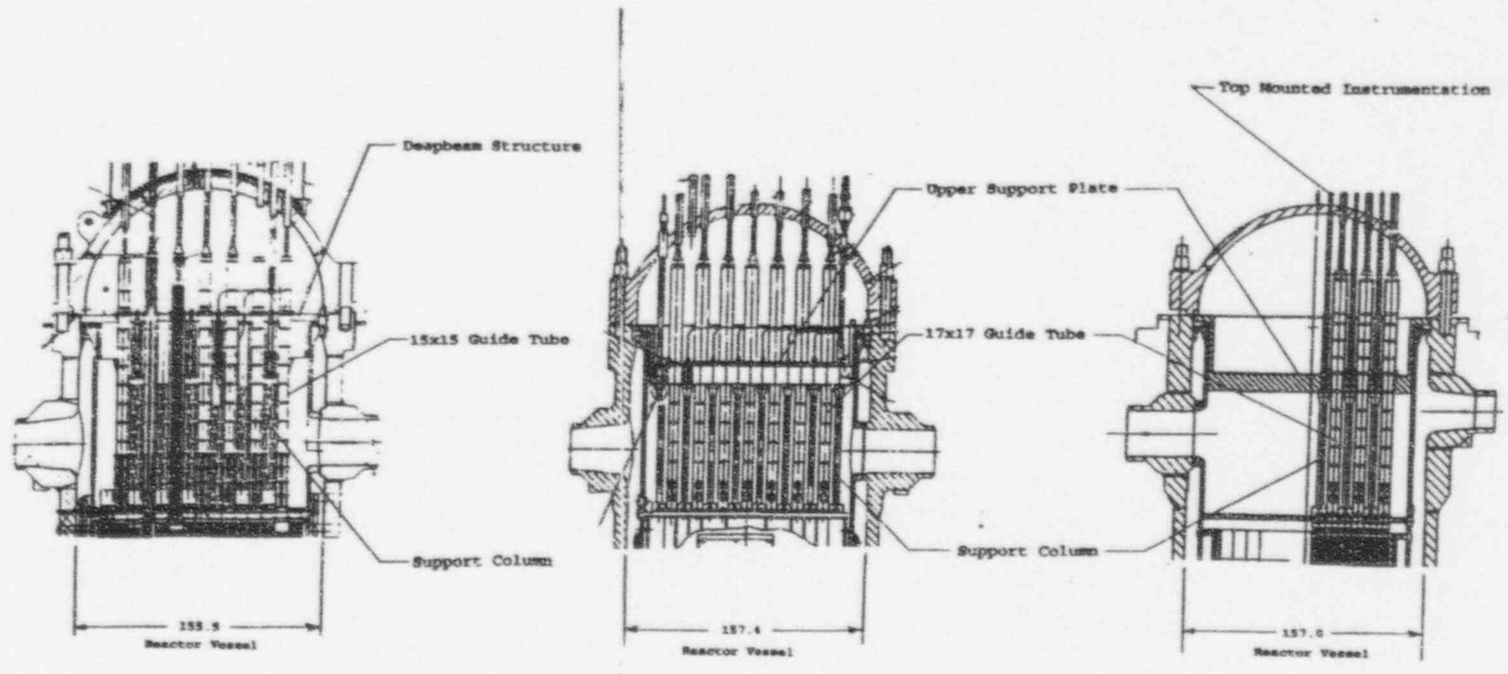
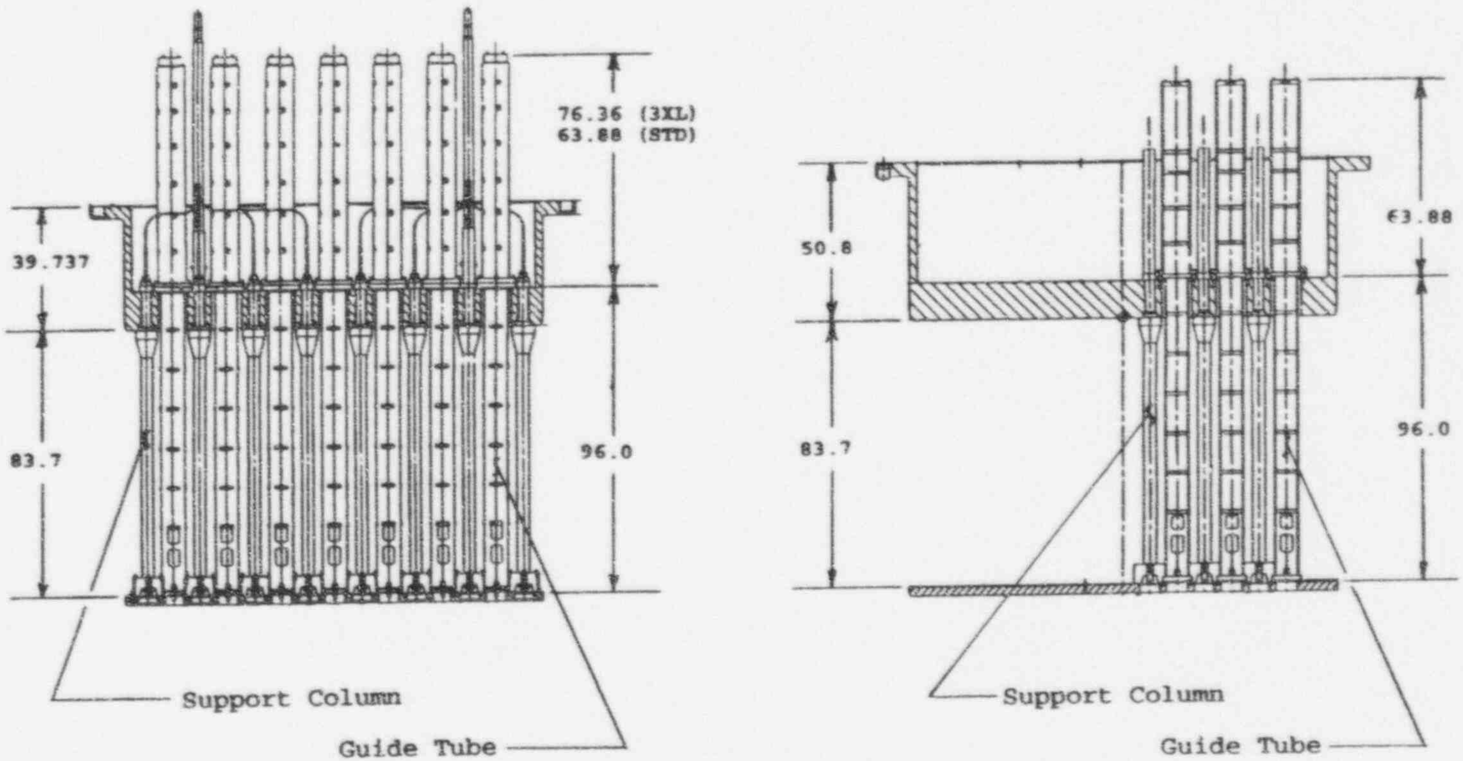


Figure 3.2 Vortex Suppression Plate



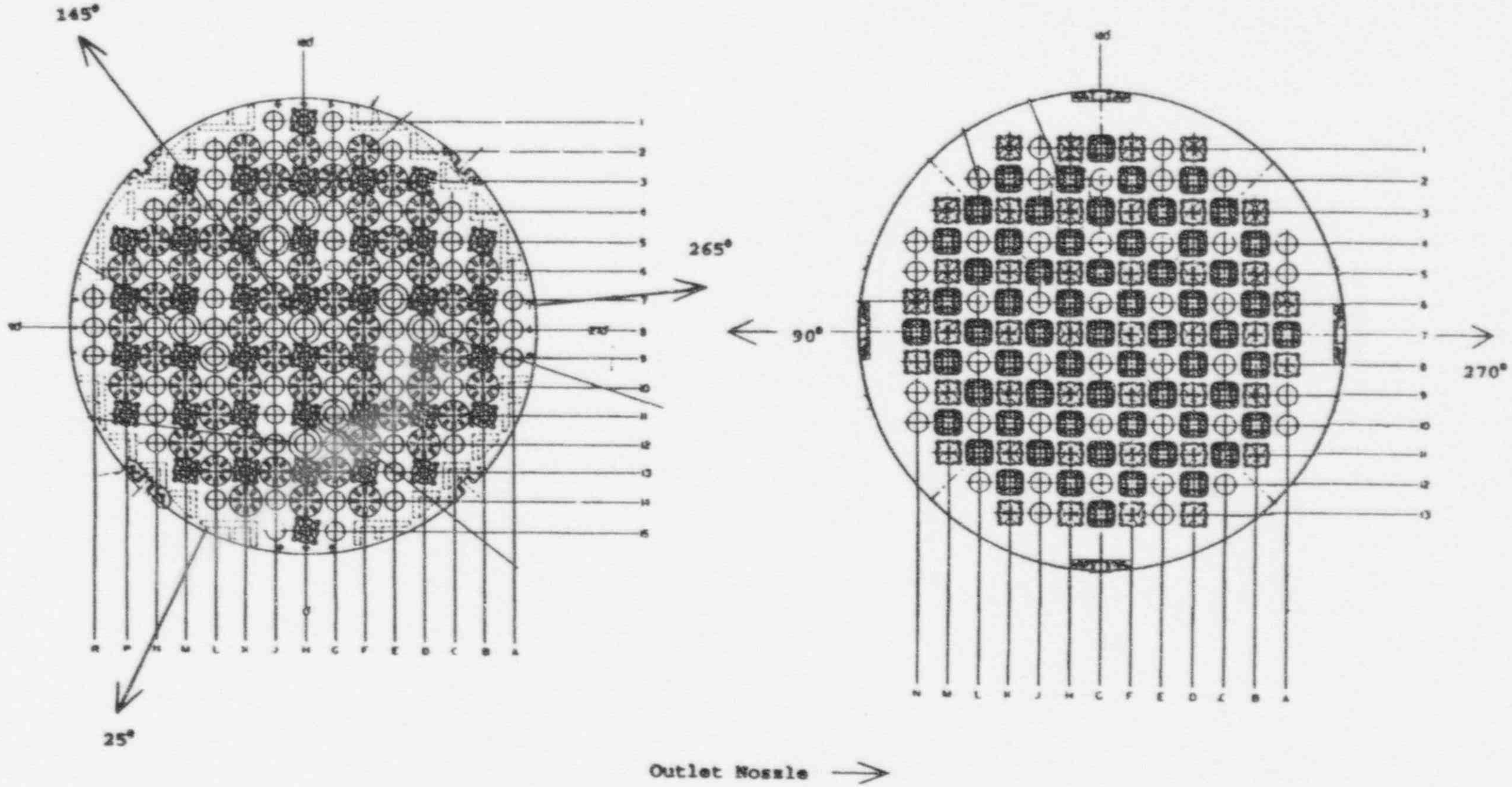
H. B. Robinson No. 2 | Doel 4/Doel 3 | AP60

Figure 3.3 Design Differences in Upper Internals



Doel 3/Doel 4 AP600

Figure 3.4 Upper Internals (Elevation)



Doel 3/Doel 4

AP600

Figure 3.5 Upper Internals (Cross-Section)

The AP600 upper internals include an extension to the support columns above the upper support plate which is part of the new top-mounted in-core instrumentation system. This tubular component is mounted to the top surface of the upper support plate and the upper end fits within a counterbore provided in the head penetration tube. Therefore, the support column extension behaves as a cantilever beam with the free end displacement limited by the clearances in the penetration. Little excitation of the upper support column extensions is expected from the [ ]^{b,c} percent core bypass flow which is directed into the reactor vessel head plenum through the head cooling nozzles. The top of the support column extensions and the head cooling nozzles are at approximately the same elevation. The bypass flow is directed vertically upward which, in combination with the lateral separation from the nearest support column extension, prevents direct impingement of the flow on the support column extension.

4.0 WESTINGHOUSE EXPERIENCE - SCALE MODEL AND PLANT TESTS

4.1 General

The program developed by Westinghouse to demonstrate and support the adequacy of reactor internals due to flow-induced vibrations includes, (a) theoretical studies to determine natural frequencies, mode shapes, vibration amplitudes and stresses, (b) scale model tests, (c) vibration measurement tests on prototype reactors, and (d) Pre- and Post-Hot Functional examination on each plant. This program which has been applied to 1, 2, 3, and 4 loop reactors, has shown that the behavior of the 4 sizes is essentially similar. Therefore, the results obtained on them has complemented one another and made possible a better understanding of the flow-induced vibration phenomena.

Table 4.1 is a summary of these tests. It provides a background for the information that is used to assess the AP600 reactor internals vibrations.

4.2 Test Results Applicable To Lower Internals

Table 4.2 summarizes information on core barrel beam mode (with no lower contact at the restraints) vibration amplitudes. These data are discussed in the following paragraphs.

- Scale model tests (Reference 3) as well as prototype tests (Reference 2) indicate that designs that have c ular thermal shields have higher beam mode vibration amplitudes than those with neutron pads. Based on scale model test data, the H. B. Robinson core barrel vibration amplitude is estimated to be [ ][b] mils rms.

- The core barrel vibration of two 3 loop neutron pad core barrels have been measured in plant tests. The amplitude measured at Tricastin, the French 3 loop prototype was the lower, possibly due to some restraint of the core barrel at the lower restraints. Estimates of the vibration amplitude from scale model tests are similar to or greater than the value measured during the Tricastin plant test. This and other scale model-prototype comparisons provide confidence in the core barrel vibration estimates from scale model data.

- The scale model estimates for Paluel vibration amplitudes are somewhat greater than the plant measurements. This indicates that removal of the neutron pads does not significantly affect the vibration amplitude. The slightly higher natural frequency resulting from reduced hydrodynamic mass and structural mass, and the lower downcomer velocity resulting from increased downcomer area (~[ ][a,b] in 3 loop plants) would tend to reduce the excitation of the core barrel as a result of removing the neutron pads.

- Estimating the AP600 core barrel vibration amplitude from scale model data results in an amplitude of approximately [ ][b] mils rms at hot functional test flow rates.

- Estimates of the AP600 core barrel cantilever beam mode vibration amplitude were obtained with the reactor system analytical model. In this model the downcomer turbulence excitation was based on 4 loop data applied to the 3 loop core barrel size. Excitations acting on the vortex suppression plate, secondary core support plate, and lower support columns as well as the other internal components and the reactor vessel were included in the model. The resulting amplitude is greater than predicted from scale model tests due to the added excitation on the lower structure. The amplitude without the excitation is somewhat less than the amplitude predicted from scale model data.

From the above, the vibration amplitude of the AP600 core barrel is estimated to be less than or equal to that of previous 3 loop core barrel vibration amplitudes. Since the core barrel flange is the same as in previous 3 loop plants, the vibration-induced stresses will be the same or less.

The core barrel natural frequencies are also shown in Table 4.2. Similar natural frequencies are shown for all data. The differences are primarily due to differences in the hydrodynamic mass generated in the downcomer annulus.

The core barrel shell mode measured responses from the 3XL scale model and from the plant test data (References 9 and 33) are summarized in Tables 4.3 and 4.4. These results are used in Section 7 to estimate the AP600 shell mode responses.

TABLE 4.1

TEST INFORMATION ON FLOW INDUCED VIBRATION OF REACTOR INTERNALS

Test	Purpose/Main Observations
• 4-Loop Thermal Shield 1/24 Scale Model	• Lower Internals Flow Induced Vibration.
	• Comparison with Prototype Data.
• 3-Loop Thermal Shield 1/22 Scale Model	• Lower Internals Flow Induced Model Test
	• Similar 3 and 4-Loop Internals Vibration.
• Indian Point Plant Test (4 Loop Prototype)	• Verified Adequacy of Internals.
	• Verified Accuracy of Scale Models and Analyses.
	• Verified Use of Hot Functional Test (No Core) for Vibration Assessment.
	• Provided Data for Improved Predictions and Forcing Functions.
• H. B. Robinson (3 Loop Prototype)	• Showed Behavior Similar to 4-Loop.
	• Provided Data for Improved Predictions.
• 4-Loop Neutron Pad 1/24 Scale Model Test	• Showed Neutron Pad Internals Have Behavior Similar to Thermal Shield Internals And Vibration Levels to be Significantly Lower.
• 3-Loop Neutron Pad 1/22 Scale Model Test	• (Same As Above).
	• Showed 3 and 4-Loop Internals Vibration Similar.
• R. E. Ginna (2 Loop Prototype)	• Two Loop Internals Vibration Similar to 3 and 4 Loop Internals Vibrations

TABLE 4.1 (Continued)

TEST INFORMATION ON FLOW INDUCED VIBRATION OF REACTOR INTERNALS

Test	Purpose/Main Observations
• Trojan-1 Plant Test (Neutron Pad Prototype)	• Additional Verification of Scale Model and Analysis Predictions.
	• Verified Adequacy of 17x17 Guide Tubes and Neutron Pad Designs.
	• Verified Reduced Vibration Levels of Neutron Pad and 17x17 Design Changes.
• 1/7 Scale UHI Internals Test	• Flow Induced Vibration of Solid Support Columns and 17x17 Guide Tubes In the UHI Array.
	• Showed, with Analysis, Adequacy of Guide Tubes, and Upper Support Columns.
• Sequoyah Plant Test (UHI Prototype)	• Verified Adequacy of Support Columns and Guide Tubes in Inverted Hat Design. Additional Verification of Scale Model Testing Technique.
• 4XL 1/7 Scale Model Test	• Show Similarity of UHI and 4XL Upper Internal Responses.
• 3XL 1/7 Scale Model Test	• Vibration Response of 3XL Internals.
	• Show Similarity of 3-Loop, 4-Loop, and UHI Upper Internals Vibratory Behavior.
• Doel 3 Plant Test (French Neutron Pad Prototype)	• Verified Adequacy of 3-Loop Inverted Hat Style Upper Internals.
• Doel 4 Plant Test (XL Prototype)	• Verified Adequacy of 3XL Lower Internals.
• Paluel Plant Test (French XL Prototype)	• Verified Lower Response of Core Barrel without Neutron Pads.

TABLE 4.2

CORE BARREL CANTILEVER BEAM MODE RESPONSE

Plant	Core Barrel Configuration	RMS Amplitude (Inches)	f_n (Hz)	Total Plant Flow-Rate (GPM)	Data Source
H.B. Robinson			[b,c]	268,500	1/22 Scale Model
Tricastin				315,600	Plant Test
				315,600	1/22 Scale Model
				315,600	1/7 Scale 3XL Model
Doel 4				322,500	Plant Test
				322,500	1/22 Scale Model
				322,500	1/7 Scale 3XL Model
South Texas (4 Loop)				426,400	1/7 Scale 4XL Model (Reference 31)
Paluel 1 (4 Loop)				442,000	Plant Test
				442,000	1/7 Scale 4XL Model (w neutron pads)
				442,000	1/24 Scale Model (w neutron pads)
AP600				212,200	w Core } System Model
				220,700	w/o Core } Analysis
				212,200	1/7 Scale Model
				220,700	1/24 Scale Model

TABLE 4.3

COMPARISONS OF SHELL MODE FREQUENCY (HZ)

Shell Modes	1/22 Scale Model 3-Loop (312)	1/7 Scale Model 3-Loop XL (314)	Tricastin 1 Plant Test	Doel 4 Plant Test
n = 2				
n = 2'				
n = 3				
n = 3'				

[b,c]

TABLE 4.4

COMPARISON OF MAXIMUM SHELL MODE DEFORMATIONS

Shell Modes	3-Loop XL Model Measurements	Tricastin 1 Measurements	Doel 4 Measurements [b,c]
n = 2			
n = 3			

ue = 1.0 x 10^{-6} in/in.

4.3 Test Data Applicable To Upper Internals

GENERAL

As mentioned earlier, the AP600 inverted top hat upper support structure is the same design as the Doel 3 and Doel 4 upper internals; and, therefore, the same general vibrational behavior is expected. Since the flow turbulences are generated by cross-flows which converge on outlet nozzles, the more highly loaded components (such as upper support columns and guide tubes) lie within the vicinity of the outlet nozzles. The AP600 outlet nozzle velocity is less than the previously tested 3 or 4 loop plants, and the corresponding upper internals flow loads are expected to be lower. Therefore, the results presented here for the AP600 upper internal components correspond to those guide tubes and support columns which are within the vicinity of outlet nozzles (see Figure 3.5).

UPPER SUPPORT PLATE

A vertically sensitive accelerometer was mounted to detect vibration at the center of the upper support plate in Doel 4 hot functional testing. The data from this accelerometer is used in the evaluation of upper support plate stress in Section 7 of this report.

GUIDE TUBES AND SUPPORT COLUMNS

In Section 7, it is shown that the mean flow induced loads (and pump-induced fluctuating loads) on the most highly loaded components are less than the loads on previously instrumented (Doel 3) and analyzed (for Sizewell B) components. The mean flow loads have been shown to be related to vibration amplitude using 1/7 scale model data (Reference 23). In the following 1/7 scale model data are compared to plant data for three loop configurations, showing the adequacy of the 1/7 scale model for prediction of upper internals component vibrations.

The 3XL 1/7-scale model test results for the guide tube (K-14) and the support column (M-3) are shown in Figures 4-1 and 4-2. The response shown in Figures 4-1 and 4-2 represents the guide tube and support column vibration spectrum (strain versus frequency) measured by axial strain gages. Tables 4-5 through 4-7 show the comparisons between the 3 loop XL 1/7-scale model test data and the Doel 3 plant data. These tables show essentially the same vibrational behavior as indicated above for the UHI plants. For example, Table 4.5 shows a good agreement in frequencies of upper internals. Tables 4-6 and 4-7 clearly show that the 1/7 scale model of 3XL plant over-predicts the response of the upper support columns and guide tubes when compared to Doel 3 plant test data. Guide tube mean flow and vibration measurement have also been made in a two loop plant (Reference 35). The results showed behavior similar to guide tubes measured in three and four loop plants.

b,c

Figure 4.1 Guide Tube Strain Gages (K-14) in Axial Direction Located at θ=270°

Figure 4.2 Axial Strain Gages at Support Column (M-3) at $\theta=90°$

b,c

TABLE 4-5

3-LOOP XL SCALE MODEL VERSUS DOEL 3
UPPER INTERNALS FREQUENCIES

Component	3XL 1/7 Scale Model	Doel 3
		b,c
Lower Guide Tube		
0° - 180°		
90° - 270°		
Upper Guide Tube		
0° - 180°		
90° - 270°		
Support Column		

*From D-Loop Test of XL G.T.

TABLE 4-6

3-LOOP XL SCALE MODEL VERSUS DOEL 3
UPPER INTERNALS STEADY FLOW LOADS

Component	3XL 1/7 Scale Model Model (LB_f) Adjusted For 3XL Flow Flow With Core	Adjusted For Doel 3 Flow	Doel 3 (LB_f) Measured During Hot Functional [b,c]
S.C. M-3			
S.C. B-7			
G.T. K-14			
G.T. B-8			
G.T. H-14			

TABLE 4-7

3-LOOP XL SCALE MODEL VERSUS DOEL 3
UPPER INTERNALS RANDOM FLOW-INDUCED VIBRATORY RESPONSE

Component	3XL 1/7 Scale Model (MILS) Adjusted For 3XL Flow	3XL 1/7 Scale Model (MILS) Adjusted For Doel 3 Flow	Doel 3 (MILS) Measured During Hot Functional [b,c]
S.C. M-3			
S.C. B-7			
G.T. K-14			
G.T. B-8			
G.T. H-14			

*Based on measured RMS strains and $\mu\varepsilon$/mil values determined from static calibration test.

5.0 RCP INDUCED PULSATIONS

The ACSTIC Computer Code (Reference 17) was used to evaluate the pump induced vibration condition for the AP-600 upper support plate, lower support plate, guide tubes, and support columns (Reference 20). The entire system model could not be evaluated due to size limitations, therefore, a one quarter symmetry reactor coolant system model was used. The model consisted of the steam generator, reactor coolant pump, hot leg, cold leg, reactor vessel, and internals. A total of 207 nodes and 331 flow paths were used in the 1/4 system ACSTIC computer model. The nodes in ACSTIC represent volumes with the node at the column center.

The model was loaded at the reactor coolant pump with [ ]^{b,c} psi and [ ]^{b,c} psi loads representative of the 29.2 Hz and 204.6 Hz forcing functions respectively. The 409.2 Hz frequency range forcing function is approximately [ ]^{b,c} psi. These levels were inferred from laboratory tests on Models 100A and 93A-1 pumps. The horsepower of the AP-600 pumps are significantly lower than the tested pumps (3000 Hp versus 8000 Hp for the Model 100A). All the ACSTIC computer runs used a [ ]^{b,c} psi load which enables the results to be easily scaled to other loading conditions.

The ACSTIC results yield radial, axial, and azimuthal gradients for the various components under investigation. The pressure gradients extracted from the ACSTIC results were tabulated for the upper and lower support plates, guide tubes and support columns areas. The values tabulated in Tables 5.1 and 5.2 represent the maximum pressure gradient for the given frequency. The maximum psi/inch loads for the guide tubes and support columns represent the SRSS of the maximum radial and azimuthal loads in the upper plenum. The guide tube and support column loads are at three elevations, (top, middle, and bottom). The elevations are at the node locations which are at the volume center. These maximum gradients were then compared to the results of reference plants in References 18 and 19.

The upper support plate and the lower support plate were evaluated at the 29.2 Hz frequency range and the guide tubes and support columns were evaluated at 204.6 Hz and 409.2 Hz frequency range. Two types of ACSTIC computer runs were used to evaluate the pump induced vibration issue; 1) RCP pumps in phase, and 2) RCP pumps 180 degrees out of phase.

Results for the upper support plate and lower support plate are listed in Table 5.1 for both cases and compared to the results of Reference 18, (3XL plant) which is of similar design and considered to have acceptable pump induced vibration levels.

The guide tube and support column results were only tabulated for the pumps out of-phase condition since this yielded the highest results. The calculated AP-600 maximum pressure gradients listed in Table 5.2 are compared to the TVA plant (Reference 19) which has similar guide tubes and support columns with acceptable pump induced vibration levels. Pressure gradients for Sizewell B, a 4 loop standard plant (i.e., not UHI) from Reference 20 are also shown in Table 5.2.

Both References 18 and 19 had actual test data and analytical models which had good correlation between the test data and calculated values, therefore the results of the ACSTIC analysis is expected to accurately predict the response in the AP-600 plant.

TABLE 5.1

ACSTIC RESULTS FOR UPPER AND LOWER SUPPORT PLATES

Component	Frequency (HZ)	Location	Calculated Maximum Difference AP-600	3XL (DOEL 4)
				b,c
Upper Support Plate	29 HZ Range Pumps in Phase	N/A		
	29 HZ Range Pumps out of Phase	N/A		
Lower Support Plate	29 HZ Range Pumps in Phase	N/A		
	29 HZ Range Pumps out of Phase	N/A		

TABLE 5.2

ACSTIC RESULTS FOR GUIDE TUBES AND SUPPORT COLUMNS

Component	Frequency (HZ)	Location	Calculated Maximum Gradient AP 600	Sequoyah 1	Sizewell B [b,c]
Guide Tubes	204 HZ Range	Top			
		Middle			
		Bottom			
	409 HZ Range	Top			
		Middle			
		Bottom			
Support Columns	204 HZ Range	Top			
		Middle			
		Bottom			
	409 HZ Range	Top			
		Middle			
		Bottom			

6.0 AP-600 REACTOR SYSTEM BEAM MODE VIBRATIONS AND FORCES

In order to determine the interaction between the AP-600 reactor vessel and internals components, an analytical model was constructed. This model shown as Figure 6.1 represented the reactor vessel, core barrel, radial reflector, and fuel assemblies as concentric beams connected at appropriate locations. The upper internals guide tube and support columns, the vortex suppression plate, and secondary core support structures are also modeled as equivalent beams. The reactor vessel support and primary loop piping stiffness at the reactor vessel nozzles are also represented in this analytical model.

The response of the AP-600 system was determined by applying forcing functions to the internals components. The forces were applied to the vessel, core barrel, core, guide tubes, support columns, the vortex suppression plate and the radial reflector. Responses were calculated for each component using the WERWOLF time integration code (Reference 34).

The AP-600 modal analyses were run both with and without the fuel assemblies in order to represent the normal operation and hot functional test plant conditions respectively. The corresponding flow rates are [ ]^{b,c} gpm at hot full power with core and [ ]^{b,c} gpm during the Hot Functional test. These analytical results indicate similar frequency responses and mode shapes with and without the core. Similar behavior has also been demonstrated in plant vibration test programs conducted at Trojan 1 and Sequoyah 1 during Hot Functional and Initial Startup testing.

The results (Reference 36) of this analytical model include the natural frequency and mode shapes of the beam responses of the reactor vessel and internals components. Figure 6.2 shows the dominant first core barrel beam mode response both with and without the fuel assemblies. The similar first mode responses of the radial reflector are shown in Figure 6.3. The natural frequencies and amplitudes are listed in Table 6.1.

The results of the dynamic analyses were provided for a variety of gap configurations at the lower radial restraints, the radial reflector pins, and at the upper core plate pins as identified in Table 6.2. From these configurations, the corresponding component moments and shear forces are given in Table 6.3, displacements given in Table 6.4, and impact forces given in Table 6.5.

The component bending moments and shear forces exhibited trends relative to the cases with or without impact occurrence at the lower radial restraints or other pin locations. For the core barrel and upper support flanges, impacting at the lower radial restraints or upper core plate alignment pins produces moments somewhat [ ]^{b,c}. Shear forces, on the other hand, become [ ]^{b,c}. At the lower support plate, both moments and shear forces [ ]^{b,c}. This is true for the interface between the core barrel and the lower support plate and between the lower support plate and the radial reflector. Only the vortex suppression plate columns exhibit [ ]^{b,c} from impacting at the lower radial restraints. This is likely due to the [ ]^{b,c} of the flow turbulence force applied to the vortex suppression plate.

Two cases were studied to determine the effect of impacting on component displacements. The response of the core barrel relative to the vessel [ ]^{b,c} at the lower radial restraints, as would be expected, and the response of the radial reflector relative to the core barrel likewise [ ]^{b,c}. The core barrel relative to the vessel displacement power spectral densities at the lower restraints are given in Figure 6.4 for the centered gap case and for the minimum 0.1 mil gap case. As seen, the core barrel cantilever response frequency [ ]^{b,c} The large harmonic response near [ ]^{b,c} is

due to a [ ]^{b,c} force on the core barrel and vessel at the pump shaft frequency. The harmonic responses at [ ]^{b,c} Hz are due to [vortex shedding]^{b,c} forces on the lower columns and vortex suppression plate. The response displacement of the upper internals relative to the core barrel is affected [ ]^{b,c} Since smaller gaps in general [ ]^{b,c} then for all cases, [ ]^{b,c} were used to produce the maximum displacements.

As shown in Figures 6.5 and 6.6, the calculated vibration response of the vortex suppression plate is overwhelmingly at [ ]^{b,c} For conservatism, [ ]^{b,c} was applied at a frequency [ ]^{b,c}

The core barrel lower radial restraints and radial reflector pin impact loads exhibited [ ]^{b,c} (see Table 6.5). With all restraints centered, the available hot gaps are [ ]^{b,c} The upper core plate alignment pins showed [ ]^{b,c} from a 0.1 to 1.0 mil minimum gap. Although not performed, it is expected that at larger gaps, [ ]^{b,c}

b,c

Figure 6.2 Mode Shapes For AP600 With And Without Core

b,c

Figure 6.3 Reflector Beam Mode With And Without Core

TABLE 6.1
AP600 ANALYSIS
(WITH AND WITHOUT CORE)

	AP600 (With Core)	AP600 (Without Core)
Fuel Assembly		
First Mode		
Natural Frequency		
Core Barrel		
Cantilever Beam Mode		
- Natural Frequency		
Radial Reflector		
Cantilever Beam Mode		
- Natural Frequency		
Reactor Vessel		
Rocking Beam Mode		
- Natural Frequency		

b,c

TABLE 6.2
PIN-CLEVIS GAP CONFIGURATIONS FOR AP600
REACTOR VESSEL SYSTEM BEAM MODEL FLOW-INDUCED VIBRATION BEAM MODEL RUNS

Interface	Minimum Initial Interface Gap (Mils)
Core Barrel - Upper Core Plate Alignment Pins	[ ][b,c]
Core Barrel - Reflector Alignment Pins	
Core Barrel - Reactor Vessel Lower Restraints	

Ctrd = Same gap on both sides. The hot single-side gaps are:

- Core Barrel - Upper Core Plate Alignment Pins [ ][b,c]
- Core Barrel - Reflector Alignment Pins [ ][b,c] inches
- Core Barrel - Reactor Vessel Lower Restraints [ ][b,c] inches

TABLE 6.3
SYSTEM MODEL VIBRATION INDUCED MOMENT AND SHEAR FORCES

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

b,c

Key

USF = Upper Support Flange
UVL = Upper Vessel Ledge

Parameter		Moment, M(in. lb.) Or Shear, S (lbf)
USF	M	
Mean		
Rms		
Max		
Min		
UVL	M	
Mean		
Rms		
Max		
Min		
USF	S	
Mean		
Rms		
Max		
Min		

b,c

TABLE 6.3
SYSTEM MODEL VIBRATION INDUCED MOMENT AND SHEAR FORCES (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

[ ] b,c

Key

UVL = Upper Vessel Ledge
CBF = Core Barrel Flange
LVL = Lower Vessel Ledge

Parameter		Moment, M(in. lb.) Or Shear, S (lb)
UVL	S	
Mean		
Rms		
Max		
Min		
CBF	M	
Mean		
Rms		
Max		
Min		
LVL	M	
Mean		
Rms		
Max		
Min		

[ ] b,c

TABLE 6.3
SYSTEM MODEL VIBRATION INDUCED MOMENT AND SHEAR FORCES (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	] b,c

Key

CBF = Core Barrel Flange
LVL = Lower Vessel Ledge

Parameter		Moment, M(in. 1b.) Or Shear, S (1b.)
CBF	S	[
Mean		
Rms		
Max		
Min		
LVL	S	
Mean		
Rms		
Max		
Min		] b,c

TABLE 6.3
SYSTEM MODEL VIBRATION INDUCED MOMENT AND SHEAR FORCES (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

[ ] b,c

Key

CB-LSP = Core Barrel-Lower Core Support Plate
RR-LSP = Lower End of the Radial Reflector-Lower Core Support Plate

Parameter		Moment, M(in. 1b.) Or Shear, S (1b.)
CB-LSP	M	
	Mean	
	Rms	
	Max	
	Min	
RR-LSP	M	
	Mean	
	Rms	
	Max	
	Min	

[ ] b,c

TABLE 6.3
SYSTEM MODEL VIBRATION INDUCED MOMENT AND SHEAR FORCES (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ] [b,c]
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

CB-LSP = Core Barrel-Lower Core Support Plate
RR-LSP = Upper End of the Radial Reflector-Lower Support Plate
VSP COL = Vortex Suppression Plate Columns (Upper End)

Parameter			Moment, M(in. 1b.) Or Shear, S (1b.)
CB-LSP	S		[ ] [b,c]
		Mean	
		Rms	
		Max	
		Min	
RR-LSP	S		
		Mean	
		Rms	
		Max	
		Min	
VSP COL	M		
		Mean	
		Rms	
		Max	
		Min	

TABLE 6.3
SYSTEM MODEL VIBRATION INDUCED MOMENT AND SHEAR FORCES (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ]b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

VSP-COL = Vortex Suppression Plate Columns (Upper End)

Parameter		Moment, M(in. lb.) Or Shear, S (lb.)
VSP COL	S	
	Mean	[ ]b,c
	Rms	
	Max	
	Min	

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[b,c]
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

GT	=	Guide Tube Mid-Elevation
USP	=	Upper Support Plate
UCP	=	Upper Core Plate
SC	=	Support Columns (In the Core Outlet Plenum)
UGT	=	Upper Guide Tube

Parameter		Displacement Ux (Inches)
GT-USP	UX	[b,c]
	Mean	
	Rms	
	Max	
	Min	
UCP-USP	UX	
	Mean	
	Rms	
	Max	
	Min	

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ]b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

GT	=	Guide Tube Mid-Elevation
USP	=	Upper Support Plate
UCP	=	Upper Core Plate
SC	=	Support Columns (In the Core Outlet Plenum)
UGT	=	Upper Guide Tube

Parameter		Displacement Ux (Inches)
SC-USP	UX	[ ]b,c
	Mean	
	Rms	
	Max	
	Min	
UCP	UX	
	Mean	
	Rms	
	Max	
	Min	

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ]b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

GT	=	Guide Tube Mid-Elevation
USP	=	Upper Support Plate
UCP	=	Upper Core Plate
SC	=	Support Columns (In the Core Outlet Plenum)
UGT	=	Upper Guide Tube

Parameter		
USP	UX	[ ]b,c
	Mean	
	Rms	
	Max	
	Min	
UGT-USP	UX	
	Mean	
	Rms	
	Max	
	Min	

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ]b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

GT = Guide Tube Mid-Elevation
USP = Upper Support Plate
UCP = Upper Core Plate
SC = Support Columns (In the Core Outlet Plenum)
UGT = Upper Guide Tube

Parameter

UCP-CB UX

Mean
Rms
Max
Min

UCP-USP UX

Mean
Rms
Max
Min

[ ]b,c

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ]b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

RR	=	Upper End of the Radial Reflector
CB1	=	Core Barrel (At UCP Elevation)
CB2	=	Core Barrel Relative To Vessel At The Elevation Of The Core Barrel Lower Restraints

Parameter

CB2-VES UX

Mean
Rms
Max
Min

RR-CB1 UX

Mean
Rms
Max
Min

[ ]b,c

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ] b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

USP = Upper Support Plate
VES = Reactor Vessel (At The Vessel Support Elevation)

Parameter

USP-VES	UX	[ ] b,c
Mean		
Rms		
Max		
Min		
VES	UX	
Mean		
Rms		
Max		
Min		

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ]b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

VSP = Vortex Suppression Plate
LSP = Lower Core Support Plate

Parameter

VSP-LSP	UX	[ ]b,c
	Mean	
	Rms	
	Max	
	Min	
LSP	UX	
	Mean	
	Rms	
	Max	
	Min	

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ] [b,c]
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

F/A = Core, At Mid Elevation
RR = Upper End of Radial Reflector

Parameter

F/A-RR	UX	[ ] [b,c]
Mean		
Rms		
Max		
Min		
F/A	UX	
Mean		
Rms		
Max		
Min		

TABLE 6.4
SYSTEM MODEL VIBRATION DISPLACEMENTS (Cont.)

Interface	Minimum Initial Gap (Mils)
Core Barrel - Upper Core Plate	[ ]b,c
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	

Key

RR = Upper End of the Radial Reflector

Parameter

Parameter	UX
RR-LSP	
Mean	[ ]b,c
Rms	
Max	
Min	

TABLE 6.5
SYSTEM MODEL VIBRATION-INDUCED PEAK INTERFACE FORCES

<u>Interface</u>	<u>Minimum Initial Interface Gap (Mils)</u>
Core Barrel - Upper Core Plate Alignment Pins	[
Core Barrel - Reflector Alignment Pins	
Core Barrel - Reactor Vessel Lower Restraints	]^{b,c}

Interface	Peak Interface Force (lb_f)
Core Barrel - Upper Core Plate	[
Core Barrel - Reflector	
Core Barrel - Reactor Vessel	]^{b,c}

\+ Maximum values calculated for a real time of [ ]^{b,c} seconds.

b,c

Figure 6.4 Lower Radial Restraint Displacement PSD
a) Centered Gaps
b) 0.1 Mil Minimum Gap

b,c

Figure 6.5 PSD of Vortex Suppression Plate Displacement Relative To The Lower Core Support Plate

Figure 6.6 PSD of Vortex Suppression Plate Displacement Relative To The Lower Core Support Plate

7.0 EVALUATION OF AP600 VIBRATION RESPONSES

7.1 Lower Internals

The AP600 lower internals core barrel assembly has the same flange dimensions, wall thickness and radii as the standard 3 loop plants (Reference 24). Both the AP600 and the 3 loop plants utilize a single integrated lower core support plate attached to the core barrel with a circumferential full penetration girth weld. There are slight dimensional differences in the lower core support plates and girth weld geometries which will be discussed in the following paragraphs.

Core Barrel And Reflector Beam Modes

As discussed in Section 4.2 the AP600 core barrel beam mode stresses are expected to be similar to or less than the stresses in previous 3 loop designs. An estimate of the AP600 core barrel stresses was made using the moment and shear forces from the reactor system model and structural models.

The reactor system beam mode analysis (**Section 6.0**) includes turbulence and periodic vortex shedding excitations acting on the vortex suppression ring that is expected to be conservative. In addition, the incorporation of the reflector results in a mode at 4.6 Hz. The reflector mode vibration level is relatively small as shown in Figure 6-4. With the vortex suppression ring and reflector effects included, however, the core barrel vibration amplitude with the core in place is still calculated to be less than 1.1 mil rms (hot functional test) which is similar to or smaller than previous plant levels.

The maximum core barrel flange beam mode moment and shear force from Table 6.3 are [ ]^{b,c} inch-lbs and [ ]^{b,c} lbs. respectively. From Reference 32, the peak beam mode stress intensity in the core barrel flange is [ ]^{b,c} psi which includes both the bending moment and shear load. A strength reduction factor of 5 was conservatively used in the analysis. The peak stress is less than the allowable alternating stress of 13,000 psi at 10^{11} cycles from the ASME Code, Appendix I, Figure I-9.2.2 (Curve C). The fatigue stress ratio which is the ratio between the calculated peak beam mode stress and allowable alternating stress is [ ]^{b,c}.

Reflector Shell Modes

Initial estimates of reflector shell mode vibration amplitudes were made (Reference 28) by scaling dynamic characteristics and responses from standard core barrel responses. The results indicate reflector shell mode vibration levels of:

	n = 2	n = 3
Pump Induced	[	]^{b,c}
Turbulence Induced		

A zero-to-peak to rms ratio of [ ]^{b,c} is included for turbulence excitation based on past experience. The stress levels from the two sources are added algebraically.

From Reference 32, the peak shell mode stresses are less than [ ]^{b,c} psi and the peak beam mode stress is [ ]^{b,c} psi. A strength reduction factor of 5 was used in the calculation of peak stress. The total shell plus beam mode stress in the reflector is less than [ ]^{b,c} psi which is less than the allowable alternating stress of 13,000 psi at 10^{11} cycles. The fatigue stress ratio is greater than [ ]^{b,c}.

In addition to the reflector, the tie rod and locating pin stresses that connect the radial reflector to the lower core support plate were determined. The peak stress in the tie rod is [ ]^{b,c} psi and in the locating pin is [ ]^{b,c} psi. These stresses are less than the allowable stress of 13,000 psi.

Core Barrel Shell Modes

The core barrel response in shell modes relative to standard plants is affected by the loss of stiffening by the baffles and formers and by the potential for higher added water mass at the reflector-core barrel interface. The effects were judged to result in a core barrel n = 2 shell mode natural frequency as low as 5 Hz. Scaling standard plant data with factors to estimate the effects of a reduced stiffness and natural frequency results in shell mode responses that are up to approximately 4 times greater than standard plant levels for the core barrel:

	n = 2	n = 3
Pump Induced	[	]^{b,c}
Turbulence Induced		

Lower Core Support Plate Axial Vibration

The evaluation of the stresses in the support plate and the core barrel-to-support plate weld includes uniform pressure differentials over the plate surface for turbulence and pump-induced pulsations. The pressure differential magnitudes were taken from earlier 3 loop designs and scaled to the AP600:

Turbulence: AP600 = $(212{,}200/322{,}500)^2$ (3XL)= (0.433) 3XL

Pump Induced: AP600 = 3XL (at 19.75 Hz)

The maximum rms displacement measurement at Doel 4 at full temperature (567°F) and flow conditions was [ ]^{b,c} mil which includes both turbulence and pump-induced vibratory responses. The corresponding pump related rms displacement measured at Doel 4 was [ ]^{b,c} mils at a center frequency of approximately 25 hz. The AP600 design is expected to have significantly lower responses than those observed at Doel 4.

The AP600 lower core support plate (LCSP) is similar to recent Westinghouse 3 loop designs like Doel 4 with slight dimensional differences. The AP600 design contains 145 fuel assemblies as compared to 157 in the standard 3 loop plants. The area beneath each fuel assembly is perforated with four holes which direct the flow into the core region. Therefore, less of the AP600 lower core plate area is perforated as compared to the 3 loop plants. These flow holes are also slightly smaller in diameter in the AP600 as compared to the 3 loop plants. The above changes tend to increase the stiffness of the AP600 lower core support plate. In contrast, the AP600 lower core plate is slightly thinner [ ]^{b,c} than the 3 loop plate which decreases the plate stiffness. However, the combined effect of the above changes is expected to have little net effect on the lower core support plate stiffness and corresponding vibrational responses.

The core barrel to lower core support plate girth weld has the same geometry but the throat dimension has been increased to [ ]^{b,c} inch in the AP600 design as compared to [ ]^{b,c} inch in the standard 3 loop plant. This change reduces the stresses in the girth weld due to the core loads being transferred to the core barrel.

The lower core support plate to core barrel weld was evaluated for the 3XL plants (Reference 18). A finite element model was constructed which included the core barrel flange, shell regions and lower core support plate. This model was subjected to a [ ]^{b,c} psi downward load on the lower support plate which resulted in a [ ]^{b,c} mil vertical displacement at the LCSP centerline and weld stress of less than [ ]^{b,c} psi. The radial displacement at the upper core plate elevation was [ ]^{b,c} mils inward. The initial 90° angle between the LCSP and core barrel wall at the girth weld location increased by [ ]^{b,c} degrees.

The results of the 3XL core barrel/lower core support plate model were used to approximate the AP600 responses (Ref. 30). The stresses in the core barrel to lower core support plate weld are dependent on the forces and moments transmitted through this joint, regardless of the loading condition. Therefore, a weld stress of [ ]^{b,c} psi will correspond to an increased angle of [ ]^{b,c} degrees whether it is due to vertical displacement of the lower core support plate or the core barrel shell mode displacements.

The lower core support plate is taken to remain horizontal for the core barrel shell modes. Therefore, the core barrel mid-elevation radial displacement will be adjusted to a vertical reference which is perpendicular to the LCSP. For the core barrel / LCSP angular deformation of [ ]^{b,c} degrees, the mid-elevation point (approximately 154 inches above the LCSP) moves radially outward [ ]^{b,c} from the perpendicular reference. The net radial displacement is the combination of [ ]^{b,c} mils inward and [ ]^{b,c} mils outward which yields [ ]^{b,c} mils outward. Therefore, the weld stress of [ ]^{b,c} corresponds to a shell mode displacement of [ ]^{b,c} mils.

The random turbulence induced and nearly sinusoidal lower support plate displacements measured during the Doel 4 (Reference 9) hot functional test were [ ]^{b,c} mils rms respectively. These rms displacements are multiplied by a factor of four for turbulence and eight for pump-induced to ratio the reference plant levels to the AP600 design levels using the results of Table 5.1. In addition, a magnification factor of 25 is applied to the pump-induced displacement to conservatively account for possible coincidence.

The corresponding values used for the AP600 are:

Turbulence: AP600 = (0.433) (3XL) = 4 (0.433)[ ]^{b,c} = [ ]^{b,c} mils 0-pk

Pump Induced: AP600 = 8 (3XL) x 25 = 8 [ ]^{b,c}(25) = [ ]^{b,c} mils 0-pk

Using the vertical lower support plate displacement to weld stress relationship of [ ]^{b,c} psi/mil yields a turbulence induced weld stress of [ ]^{b,c} psi and pump-induced weld stress of [ ]^{b,c} psi.

The n = 2 and n = 3 shell mode turbulence induced responses of [ ]^{b,c} mils 0-pk are combined by the square root of the sum of the squares method to yield a displacement of [ ]^{b,c} mils. The corresponding weld stress is determined from [ ]^{b,c} psi. The pump related n = 3 shell mode weld stress is given by [ ]^{b,c} psi.

The maximum moment and shear force at the weld location are [ ]^{b,c} in.-lbs. and [ ]^{b,c} lbs. respectively as determined from the system model (Section 6.0). The maximum moment was taken as the average of the absolute values of the positive and negative moments listed in Table 6.3. The corresponding bending and shear stresses in the weld are [ ]^{b,c} psi respectively.

The total combined stress in the weld is [ ]^{b,c} psi which is much lower than the ASME Code allowable of 13,000 psi. The fatigue stress ratio of [ ]^{b,c} more than compensates for any inaccuracy involved in using the 3XL model for the AP600 responses.

Vortex Suppression Plate and Secondary Core Support

The Vortex Suppression Plate has a larger surface area and more closely approaches the lower reactor vessel head than the tie plate that it replaced. Estimates of turbulence excitation forces and forces due to the postulated periodic shedding of vortices were calculated in a manner intended to be

conservative. The resulting forces were included in the excitations applied to the reactor vessel system model discussed in Section 6.0.

The maximum calculated vibratory moment and shear force acting on the vortex suppression plate and secondary core support columns (Table 6-3) are [ ]^{b,c} in. lbs and [ ]^{b,c} lbs., respectively. The resulting alternating peak stresses documented in Reference 32 are [ ]^{b,c} psi in the inner column and [ ]^{b,c} psi in the outer column. A strength reduction factor of [ ]^{b,c} is used in calculating the peak stress. The alternating bolt peak stresses are [ ]^{b,c} psi for the inner column bolts and [ ]^{b,c} psi for the outer column bolts. A strength reduction factor of k = [ ]^{b,c} was conservatively used in determining the peak stress in the bolt. The vortex suppression plate alternating stresses in the columns and bolts are less than the allowable alternating stress of 13,000 psi at 10^{11} cycles from the ASME Code Figure I-9.2.2 Curve C. The fatigue stress ratio for the inner and outer columns are [ ]^{b,c} respectively, and for the inner and outer column bolts it is [ ]^{b,c} respectively. The fatigue stress ratio is defined as the ratio of the allowable alternating stress divided by the calculated alternating stress.

7.2 Upper Internals

Upper Support Plate and Support Structure

The pump pulsation levels and turbulence induced levels of excitation acting on the upper support plate are expected to be lower than those on Doel 4. The displacements of the AP600 upper support is estimated from Doel 4 measurements. For conservatism, the pump pulsation vibration amplitude measured at Doel 4 is multiplied by [ ]^{b,c} to account for increased coincidence with an internals vertical natural frequency and then added to the broad band (6 to 98 Hz) displacement measured on Doel 4. The AP600 upper support plate displacement estimated in this way is:

Δ USP = [ ]^{b,c} mils 0-pk

where:

- [ ]^{b,c} is the rms random response measured on Doel 4 (mils)
- 4 is a factor applied to random rms levels to obtain zero-to-peak levels
- [ ]^{b,c} is the pump pulsation level measured in Doel 4 (mils)
- [ ]^{b,c} is a factor to account for increased coincidence
- 4 is a factor applied to pump rms levels to ratio reference plant levels to those of the AP600 design using the results of Table 5.1.

Upper Support Plate and Skirt Transition

The 0 to peak vibration displacement of the center of the upper support plate is calculated to be [ ]^{b,c} inch. The displacement of [ ]^{b,c} inch is calculated to result in an equivalent solid plate stress intensity of [ ]^{b,c} psi from Reference 32. The perforated plate stress intensity at the center of the plate is calculated to be [ ]^{b,c} psi. This is based on a ligament efficiency of h/P = [ ]^{b,c} and a conservative stress multiplier of k = [ ]^{b,c}. The product of the inverse ligament efficiency, P/h, and stress multiplier, K, is multiplied by the solid plate stress intensity to arrive at the perforated plate peak stress intensity. The resulting alternating peak stress of [ ]^{b,c} psi is well below the allowable alternating stress of 13,000 psi at 10^{11} cycles from the ASME Code Figure I-9.2.2 Curve C.

At the transition locations of the skirt, the support plate to skirt radius peak stress is calculated to be [ ]^{b,c} psi and the skirt to flange radius peak stress is calculated as [ ]^{b,c} psi, from Reference 32. These locations are bounded by the center of the perforated upper support plate alternating stress of [ ]^{b,c} psi.

Guide Tubes and Support Columns

The guide tubes in the AP600 upper internals are very similar to the 17x17AS guide tubes and the support columns are identical to those in standard plants such as Doel 3 so that flow induced vibration and load correlations can be confidentially applied to the AP600 components. As shown in Table 7.1, steady flow loads predicted from 3XL 1/7 scale model data were somewhat greater than those measured in Doel 3, demonstrating the conservatism of the model results. Model results have been used to correlate the steady load calculated in the UPPLEN Code to vibratory amplitudes and shows that vibratory amplitudes are proportional to steady loads raised to a power less than 2 (Reference 6). The UPPLEN loads for Doel 3 and the AP600 are also shown in Table 7.1.

The AP600 upper internals have slightly more guide tubes and support columns than the Doel 3 plant as discussed in Section 3.1. The UPPLEN models include the AP600 number of guide tubes and support columns as well as the AP600 outlet nozzle orientation.

The UPPLEN forces calculated in the AP600 plant are less than the corresponding Doel 3 and Sizewell B forces. In addition, the Doel 3 values in Table 7.1 (and similar data for Sequoyah 1), indicates that UPPLEN and 1/7 scale model results predict higher mean flow loads than measured during preoperational plant test programs. Therefore, the actual loads on the AP600 upper internals components are expected to be lower than the UPPLEN values listed in Table 7.1.

The adequacy of the AP600 guide tubes has been demonstrated from the Doel 3 plant measurement program and for Sizewell B in References 10 and 22 respectively.

Guide tube natural frequencies occur at frequencies close to the first and second harmonics of the reactor coolant pump blade passing frequencies (see Figure 7.1) in the AP600 design. Although greater separation of these frequencies is desired it is noted that:

- analysis of guide tubes for other plants made for similar differences between pump blade passing (Ref. 37) frequencies and guide tube natural frequency have shown that the resulting guide tube stresses are well within allowable values.

- modifications to the current guide tube design are planned which may alter the calculated natural frequencies. This issue will be finalized when the pump pulsation test data have been received, and guide tube design details have been finalized.

Pump pulsation levels at the pump rotating speed and 1st blade passing frequency in the AP600 plant are calculated to be similar to those in Sizewell B (Table 5.2) for the same pump model pressure fluctuation. However, the guide tube natural frequencies are close to the first and second harmonics of the RCP blade passing frequency as shown in Figure 7.1. The coincidence are associated responses will be addressed fully when the guide tube detailed design has been finalized. In addition, guide tube responses will be measured during hot functional testing.

Since both turbulent loads are lower and pump-induced loads are similar on the AP600 guide tubes and support columns, their adequacy for the AP600 design is demonstrated by the analysis performed for the corresponding Sizewell B plant components. These components were shown to be adequate for Sizewell B in Reference 22.

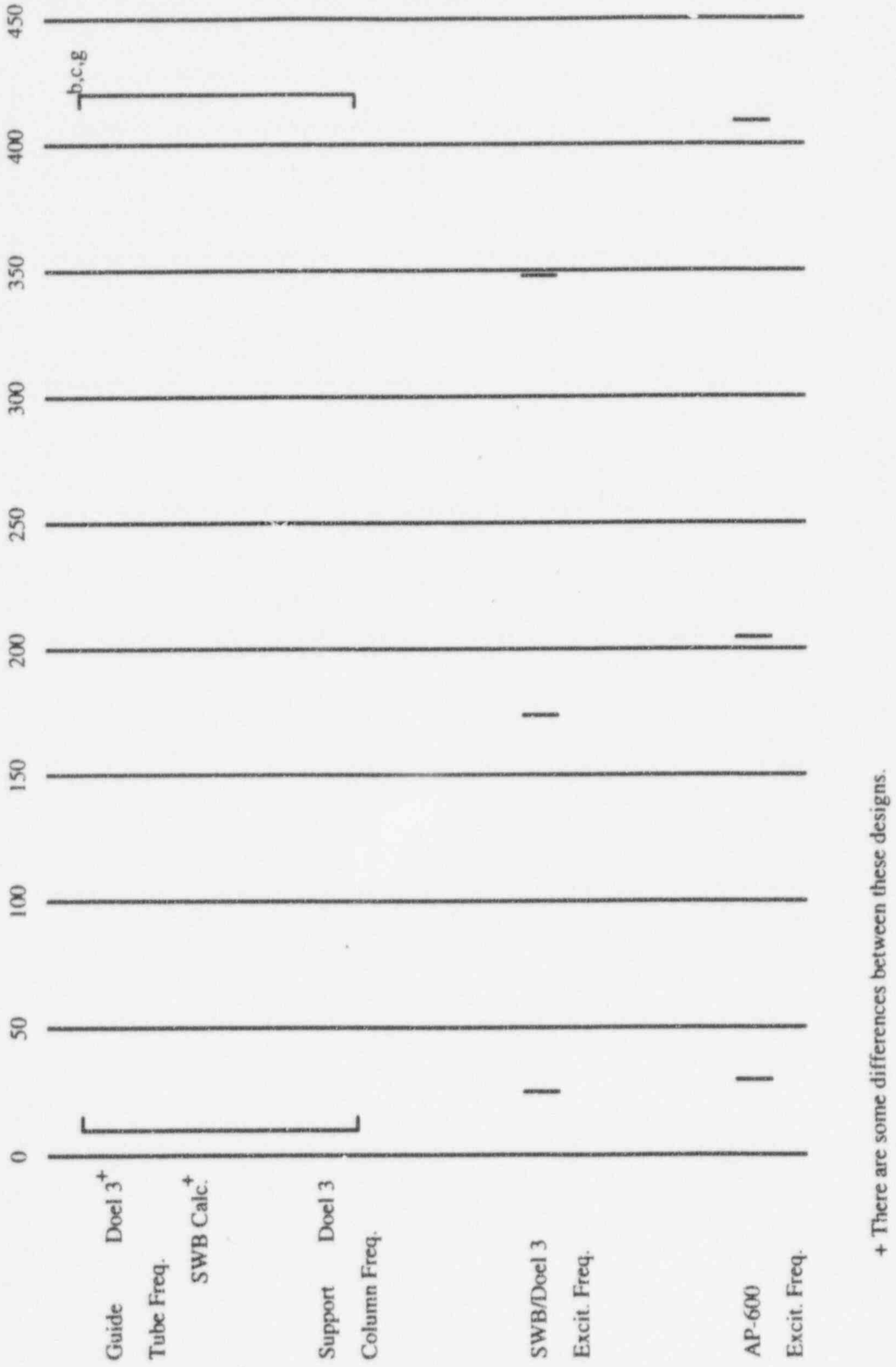


Figure 7.1 Guide Tube and Support Column Natural Frequencies and RCP Excitation Frequencies

TABLE 7.1
MAXIMUM STEADY FLOW LOADS ON UPPER INTERNALS COMPONENTS

	UPPLEN	DOEL 3 1/7 SCALE MODEL	HFT MSMTS.	AP600 UPPLEN CALC.	SWB UPPLEN CALC
Reference	23	8	8	21	22
Guide Tube					
Location	K-14	K-14	K-14	G-13	D-14
Load (lbs)	[		]^{b,c}	[ ]^{b,c}	[ ]^{b,c}
Support Column					
Location	M-3	M-3	M-3	H-13	C-14
Load (lbs)	[		]^{b,c}	[ ]^{b,c}	[ ]^{b,c}

7.3 Interface Loads

The loads due to vibration transmitted between the 3 sets of keys and clevises are listed in Table 6.5. It should be noted that the system model did not include torsional degrees of freedom (since it was expected that adequately low loads would result without torsional degrees of freedom). Calculations of this type for other designs have shown that loads are reduced significantly when torsional responses are included.

The reactor vessel clevis and the core barrel lower restraint key design specification (Reference 25) include an allowance for a vibratory load of [ ]^{b,c} lbs. From Table 6.5 the vibration induced loads at these interfaces are conservatively bounded by the peak interface load of [ ]^{b,c} lbs at the Core-Barrel - Reactor Vessel interface. This is less than the allowable vibratory load and is acceptable.

The upper core plate alignment pins are unchanged from the design in standard plants and have a [ ]^{b,c} lbs vibratory allowance (Reference 24). The reflector alignment pins utilized the same design but are slightly larger than the core plate alignment pins and have a higher load capacity. The vibration induced loads in these interfaces are conservative bounded by the peak interface load of [ ]^{b,c} lbs. in Table 6.5 at the Core Barrel - Upper Core Plate interface and [ ]^{b,c} lbs at the Core barrel - Reflector interface. These loads are well below the vibratory allowance load of [ ]^{b,c} lbs.

8.0 PRE-OPERATIONAL INTERNALS VIBRATION MEASUREMENT PROGRAM

Westinghouse is planning a reactor internals flow-induced vibration measurement program during preoperational tests of the first AP600 plant. This test will be similar to previous plant tests and will provide added assurance of the adequacy of the reactor internals design, independent of the vibration assessment program.

The major structural components of the AP600 reactor lower internals will be instrumented during pre-operational testing. Transducers will be installed on the reactor vessel and the internals prior to the Cold Hydrostatic test. The integrity of these transducers and the operability of the data acquisition equipment will be verified during this test.

The response of the reactor and the internals due to flow-induced vibration will be measured during the Hot Functional test. As shown by the results in Section 6.0, the dominant vibration modes of the internals with no core present are similar to those with the core in place and their vibration amplitudes are expected to be more than 10% higher.

Data will be acquired at several temperatures from cold startup to hot standby [ ][b,c] conditions. Data will be recorded for pump startup and shutdown transients as well as for all possible combinations of steady-state pump operation. In addition, data will be recorded with none of the pumps operating in order to determine the background noise level.

Transducer signals will be monitored as they are being recorded to insure the validity of the data. A spectrum analyzer will be used during the test as an additional check on transducer performance. The spectrum analyzer will also provide preliminary information on the natural frequencies and responses of the instrumented components. The majority of the data, however, will be analyzed from the magnetic tapes.

The leads for these internally mounted transducers will be routed through the top mounted instrumentation guide tube conduits. The combined in-core detectors/core exit thermocouples will not be installed during the Hot Functional test. Special fittings, designed to ASME Section III, Class 1 pressure boundary rules, will be used to seal the transducer leads during this test. These fittings will be removed following the test.

All transducers and associated hardware will be removed after the completion of the Hot Functional testing.

8.1 Location of Transducers

The location and types of transducers to be used for the AP600 vibration measurement program are shown in Figure 8.1. Detailed transducer locations and their directions of sensitivity are listed in Table 8.1. The measurement objectives for the instrumented components are listed below:

1. Four radially sensitive accelerometers mounted near the top of the radial reflector. These transducers are to detect shell mode vibration of the radial reflector and provide additional information on the core barrel beam modes.

2. Six axially sensitive strain gages mounted just below the core barrel flange. These transducers will detect axial vibration of the lower internals and core barrel beam modes.

3. Two axially sensitive strain gages (one inside and one outside) mounted on the upper support assembly skirt to detect vertical motion of the upper support structure or alternatively, this information may be obtained using axially sensitive accelerometers.

4. Four axially (2 inside and 2 outside) sensitive strain gages located on the core barrel to lower core plate weld. These strain gages will provide direct information on the stresses at this location or alternatively, this information may be obtained using axially sensitive accelerometers.

5. Four axially sensitive strain gages mounted on two lower support columns that attach the vortex suppression plate to the lower support plate. These gages will be mounted at 90° separation on two different support columns such that lateral displacement of the vortex suppression plate assembly can be determined. Alternatively, four horizontally sensitive accelerometers will be considered to obtain this information.

6. Two axially sensitive strain gages located on the upper support column extension. These transducers will detect the lateral displacement of the extension.

7. Four vertically sensitive and two horizontally sensitive accelerometers mounted on the reactor head closure studs at 90° intervals. These transducers will detect motion of the reactor vessel and the upper and lower internals flanges.

8. Four radially sensitive accelerometers will be installed at the upper core plate elevation to determine the shell mode responses of the core barrel.

9. Four axially sensitive strain gages will be installed near the top of the guide tube subjected to the highest flow loads. These transducers will determine the beam made displacement and response frequencies of the guide tube.

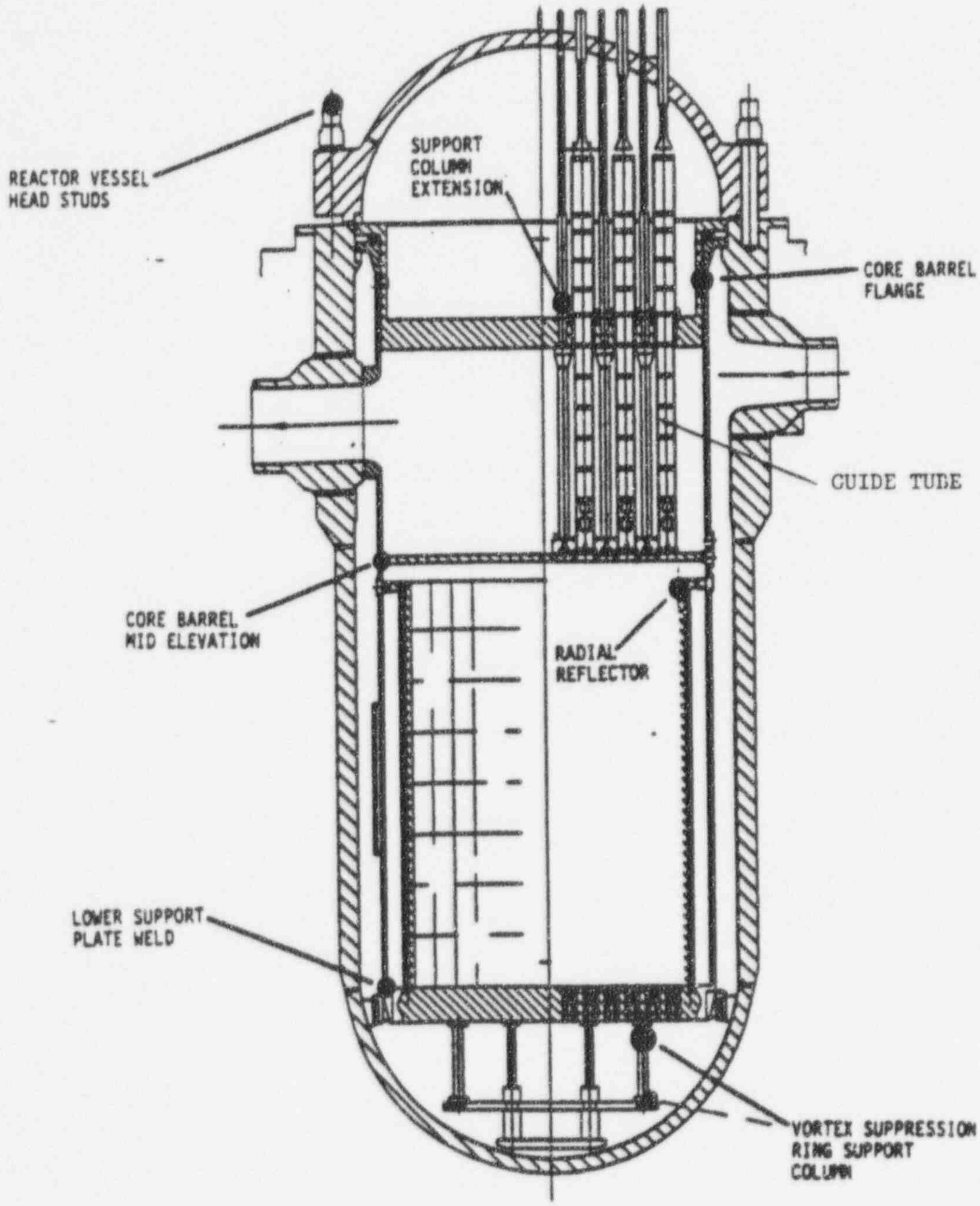


Figure 8.1 **Location of Transducers for the Preoperational Vibration Measurement Program**

TABLE 8.1 AP600 TRANSDUCER LOCATIONS			
Instrumented Component	**Number and Type of Transducers**	**Transducer Locations**	**Direction of Sensitivity**
Radial Reflector (Inner Wall)	4 Accelerometers	0°, 180°, 225°, 270°	Radial
Core Barrel Flange (Outer Wall)	4 Strain Gages	0°, 90°, 180°, 270°	Axial
Core Barrel Flange (Inner Wall)	2 Strain Gages	180°, 270°	Axial
Core Barrel Mid-Elevation	4 Accelerometers	0°, 180°, 225°, 270°	Radial
Upper Support Skirt (Inside and Outside)	2 Strain Gages	180°	Axial
Lower Support Plate Weld (Inside and Outside)	4 Strain Gages	0°, 90°	Vertical
Vortex Suppression Plate Support Columns (2)	4 Strain Gages or 4 Accelerometers	On column nears LCSP or on Vortex Suppress Ring	Axial Horizonal
Reactor Vessel (Head Studs)	4 Accelerometers 2 Accelerometers	0°, 90°, 180°, 270° 0°, 90°	Vertical Horizontal
Support Column Extension	2 Strain Gages	0°, 90°	Axial
Guide Tube	4 Strain Gages	0°, 90°, 180°, 270°	Axial
Total Number of Transducers	36		

8.2 Transducers And Data Acquisition Equipment

The strain measurements of the upper support plate skirt, lower core support plate weld, and core barrel flange will utilize weldable strain gages. These gages have a Ni-Cr filament in a 120 ohm quarter bridge configuration with a nominal gage factor of 1.90. These gages will be temperature compensated in the range of 75°F to 600°F for mounting on stainless steel. Each gage will be hydrotested by the manufacturer and helium leak checked upon receipt to verify the integrity of the stainless steel sheath.

The gages will be attached to the component being measured by a number of 5 to 10 watt-second electrical resistance spot welds. The lead wires are fiberglass insulated and enclosed in a stainless steel sheath that protects the leads until they exit the pressure vessel through an instrument penetration. Outside of the reactor vessel, the leads will be connected to bridge completion networks containing precision resistors. The strain gage bridges will be located as near to the gage as possible to minimize picking up electrical noise in the gage lead and to minimize gage desensitization. Preamplification will consist of signal conditioning amplifiers which will provide bridge balancing, shunt calibration and output filtering.

The internal accelerometers mounted on the lower internals are of the high temperature and pressure piezoelectric type. These transducers will be mounted to the instrumented components by bolts which will be secured by locking devices. These uniaxial accelerometers will be supplied with integral leads of sufficient length to exit the reactor vessel via the head penetrations.

In addition uniaxial sensitive piezoelectric accelerometers will be mounted on the reactor vessel head studs to measure horizontal and vertical movement. Short lengths of high temperature, low noise cable will extend the leads through the vessel insulation.

Both the internally and externally installed accelerometers will use fixed gain remote charge amplifiers which result in output in voltage. Power for these remote charge amplifiers will be provided by signal conditioners located near the recording equipment.

The accelerometers will be checked with a vibration shaker which provides a known acceleration level. This acceleration level will be applied to the accelerometer near its installation location and read out on the data acquisition equipment. This will act as a check on the amplifiers as well as on the accelerometers themselves.

The internally mounted transducers will have their leads protected from impinging flow utilizing methods employed in previous Westinghouse plant tests. The leads from the transducers installed on the lower internals will be shielded from flow by protective covers similar to ones used at Doel 4.

The output from the strain gage and accelerometer signal conditioners will be routed to a switching panel. Selected signals will then be further amplified and filtered by differential DC amplifiers (or equivalent) before being recorded on magnetic tape. Both input and output signals from the tape recorders will be monitored with a dual-channel oscilloscope to assure data are being properly recorded on tape without saturation and with the optimum signal-to-noise ratio.

The data will be suitably recorded to cover the 0 to 1000 Hz frequency band of interest. Recordings for steady pump operation will be 10 to 20 minutes long to allow for proper signal averaging in later analysis. For each data record, the time, tape number, tape position, recording parameters, signal conditioner parameters and pertinent reactor and test conditions will be recorded on log sheets. Onsite analysis will consist mainly of using a real-time spectrum analyzer to generate linear voltage spectra.

8.3 Equipment Calibration

The procedure to be used to ensure that all equipment is calibrated and properly functioning is as follows. Prior to shipment of the equipment to the plant site, all instrumentation will be calibrated and certified by Westinghouse or by a suitable calibration laboratory with NIST (National Institute of Standards and Technology) traceability. Upon arrival at the site, the instruments will be checked with a reference voltmeter to verify that all Westinghouse measurement equipment is functional.

At the beginning of each magnetic tape, a reference 1-volt rms 100 Hz sine wave will be inserted through the amplifiers and recorded on each data channel. This signal will be used to adjust the playback electronics during subsequent data analysis. Also recorded on tape will be signal outputs of the strain measuring system when a strain response is simulated by use of the shunt resistors in the strain gage signal conditioners. A random noise signal consisting of white noise will also be recorded on the first tape of each tape recorder in order to check the inter-channel phase relationship.

At completion of the data collection program, the measurement instruments will again be rechecked by comparing the equipment output signals to known input reference voltages.

8.4 Data Reduction

The techniques used to reduce the signals from the magnetic tapes will include generation of time histories using a recording oscillograph, and preparation of linear frequency spectra using a real time Fast Fourier Transform Analyzer. The plotted spectra will be used to determine the predominant responses and their associated frequencies. The time histories will be used to determine the change in mean response levels due to pump startup and shutdown transients.

8.5 Cold Hydrostatic Test Conditions

Data will be recorded during the Cold Hydrostatic test to assure that all transducers, instrumentation, cabling and data acquisition equipment are functioning properly. In addition, any convenient low temperature data will be acquired during pump transients and steady state operation. Signal and noise levels measured during this test will help determine the electrical gain settings to be used in recording the Hot Functional test data.

8.6 Hot Functional Test Conditions

The majority of the data will be acquired during the Hot Functional test. Data will be recorded at several temperature plateaus corresponding to the plant startup Pre-Operational test requirements. These measurements will indicate the variation in response due to temperature changes from cold startup to hot standby conditions. Maximum levels are expected to occur at 200 to 250°F. The levels in this temperature band are expected to be 20 to 25% greater than the hot full power levels. Most of the data will be recorded near the hot standby temperature of 529°F and will include pump transients as well as all possible combinations of steady-state pump operation.

The pump operating modes at which data will be recorded include:

1. No pumps operating - records made of background noise.

2. Startup transients - to record mean strains and transient vibration behavior that results from pump startup.

3. Steady operation of one or more pumps - to identify vibration response of instrumented components during various flow conditions.

4. Shutdown transients - flow induced mean strains and transient vibration behavior are deduced from these records by noting the change in strain levels through the pump transient.

A simultaneous shutdown of all four pumps may also be conducted in order to provide a direct measurement of the mean strains between full flow and zero flow conditions.

The reactor internals will be subjected to higher flow loads during Hot Functional test than for normal operation with the core installed. Vibration levels are generally lower with the addition of the core than for the Hot Functional testing (References 5 and 7). Therefore, the responses measured during the Hot Functional test will be conservative with respect to normal operating conditions.

8.7 Predicted Responses

The predicted responses will be established at a later date. This analysis will be used to determine the expected responses at the appropriate transducer locations along with corresponding acceptable values. Much of the basis for the expected values is included in this report. Acceptable values will be related to allowable values in the ASME Boiler and Pressure Vessel Code. The results of previous scale model and full scale plant tests may be incorporated in this analysis.

8.8 Test Program Summary

The adequacy of the internals design with respect to flow-induced vibration will be confirmed by the visual Pre- and Post-Hot Functional inspections. At the completion of the Hot Functional test, the internals will have been subjected to more than 240 hours of greater than normal full flow conditions. Additional time will have been accumulated with one, two, or three pumps in operation. This results in more than one million cycles of vibration for the major structural components.

The AP600 vibration measurement program will determine the natural frequencies, modes and amplitudes of the major structural components due to flow-induced vibration. These measured responses will serve as an additional technical basis to the vibration assessment program and previous plant and scale model tests.

All of the transducers and associated hardware will be removed after the completion of the Hot Functional test prior to core loading. The special pressure boundary fittings at the top of the in-core instrumentation conduits will also be removed.

The best estimate flow rates calculated with and without the fuel assemblies installed are 204,000 and 217,600 gpm respectively (References 26 and 27). Therefore, the Hot Functional test results will be conservative with respect to operation with the core installed.

9.0 PRE- AND POST-HOT FUNCTIONAL INSPECTION

The Westinghouse 3-Loop Internals Assurance Program (Reference 1) includes visual examinations of the reactor vessel and internals prior to and after the completion of the Hot Functional Test. The internals are removed from the reactor vessel and the following areas are inspected under 5X or 10X magnification:

1. All major load-bearing elements of the reactor internals relied upon to retain the core support structure in position.

2. The lateral, vertical and torsional restraints provided within the vessel.

3. Those locking and bolting components whose failure could adversely affect the structural integrity of the reactor internals.

4. Those surfaces that are known to be or may become contact surfaces during operation.

5. Those critical locations on the reactor internal components as identified by the vibration analysis.

6. The interior of the reactor vessel for evidence of loose parts or foreign material.

The results of these inspections will be recorded on the AP600 Vibrational Check-Out Functional Test Inspection Data drawing.

Acceptance Standards are the same as required in the shop by the original design drawings and specifications.

During the Hot Functional test, the internals will be subjected to a total operating time at greater than normal full flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately one million cycles on the main structural elements of the internals. In addition, there will be some operating time with one, two and three pumps operating.

When no signs of abnormal wear, no harmful vibrations are detected, or no apparent structural changes take place, the AP600 reactor internals are considered to be structurally adequate and sound for operations.

10.0 CONCLUSIONS

Vibrational test data obtained from the scale model tests and the instrumented plant tests (summarized in Tables 4.1 through 4.7) show that the internals vibration levels are low and that the AP600 reactor internals design is adequate to assure structural integrity against flow-induced vibrations. The comparison of Tables 4.1 through 4.7 also show that the AP600 reactor vessel internals design is adequately represented by the Doel 4 lower internals and the Doel 3 upper internals; and hence, the in-plant testing results obtained at those and similar plants are applicable to the AP600. The analytical results of Section 5.0 for the AP600 plant show that the expected vibrational responses are consistent with the levels measured during these previous plant tests.

The recommendations of Regulatory Guide 1.20 are satisfied by conducting the confirmatory Pre- and Post-Hot Functional visual and non-destructive surface examinations and the limited measurement program on the AP600 prototype plant. The plant preoperational test program and the Pre- and Post-Hot Functional test examinations are described in Sections 8.0 and 9.0 respectively.

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APPENDIX A

SUMMARY OF CALCULATED AND ALLOWABLE STRESSES

The following information is provided for additional information related to reactor vessel internals vibrations.

The calculated stresses or loads due to flow-induced and pump-induced vibrations and the design allowable limits are summarized in Table 1 (attached). This table shows that all calculated values are lower than allowable values.

The calculated stresses and loads are judged to include considerable conservatism. In addition to the use of mechanical design flow rates which are greater than best estimate flow rates, examples of conservatism or worst case assumptions include:

- That organized vortex streets can be shed due to crossflow on the vortex suppression plate structures and that the vortex shedding in the lateral direction will be at a frequency that is equal to the natural frequency of the structure.
- That the natural frequency of a structural mode will be coincident with pump-induced natural frequencies although coincidence is not calculated.
- The structural models include no torsional degrees of freedom of the upper and lower internals assemblies. This, in conjunction with the assumption that the loads are taken by one key/clevis at the worst case offset, results in a conservatism.

The allowable values are ASME Code fatigue design values at 10" cycles which in themselves include a factor of safety of 2 on stress.

The key and pin allowable loads are interface values which reflect the relatively small portion of the total load capability allowed for flow-induced vibration (as opposed to seismic loads, for example).

These values would be suitable for conservative expected values for plan testing. They will be considered in the evaluations carried out for the internals stress report.

Regarding the guide tubes and support columns, as discussed in the meeting, further work is required to define mains for flow-induced vibration.

SUMMARY OF STRESSES/LOADS AND ALLOWABLE VALUES				
AP600 Reactor Internals Component	**Report Page**	**Stress/Load** b,c	**Allowable***	**F. S.** b,c
Core Barrel Flange (beam mode)	53		13,000 psi(1)	
Radial Reflector (shell mode) (beam mode) (combined)	53		 13,000 psi(1) 13,000 psi(1) 13,000 psi(1)	
Reflector Tie Rod	53		13,000 psi(1)	
Reflector Locating Pin	53		13,000 psi(1)	
Lower Core Plate to Core Barrel Weld (total)	55		13,000 psi(1)	
Vortex Suppression Plate Inner Column	56		13,000 psi(1)	
Vortex Suppression Plate Inner Column Bolt	56		13,000 psi(1)	
Vortex Suppression Plate Outer Column	56		13,000 psi(1)	
Vortex Suppression Plate Outer Column Bolt	56		13,000 psi(1)	
Upper Support Plate (perforated region)	56		13,000 psi(1)	
Upper Support Plate to Skirt Radius	57		13,000 psi(1)	
Upper Support Flange to Skirt Radius	57		13,000 psi(1)	
Guide Tubes	57		13,000 psi(1)	
Support Columns	57		13,000 psi(1)	
Lower Radial Support Keys (vibratory)	60		25,000 lb(2)	
Upper Core Plate Alignment Pins (vibratory)	60		16,000 lb(3)	

Notes:

1. Based on ASME B&PV Code Section III, Appendix I, Figure I-9.2.2 (Curve C) for 10" cycles.
2. AP600 Document No. RXS-M8-001, Revision B.
3. AP600 Document No. MI01-M2C-001.
4. Westinghouse Pensacola to confirm later by analyses or test.